Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381-2000

APR 20 2004

10 CFR 50.71(e)

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

| | | |
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| In the Matter of | ) | Docket No. 50-390 |
| Tennessee Valley Authority | ) | |

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1 - CHANGES MADE TO THE WBN TECHNICAL SPECIFICATION BASES AND TECHNICAL REQUIREMENTS MANUAL

The purpose of this letter is to provide the NRC with copies of changes to the WBN Technical Specification Bases (TS Bases), through Revision 65, and WBN Technical Requirements Manual (TRM), through Revision 33, in accordance with WBN TS Section 5.6, "TS Bases Control Program," and WBN TRM Section 5.1, "Technical Requirements Control Program," respectively. These changes have been implemented at WBN during the period since WBN's last update (September 19, 2002) and meet criteria described within the above control programs for which prior NRC approval is not required. Both control programs require such changes to be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). WBN's FSAR update in accordance with 10 CFR 50.71(e) is being provided under separate cover.

Enclosure 1 provides the changes made to the TS Bases for this period. Note that changes made to the TS Bases under approved license amendments to the WBN TS are not included unless necessary for page integrity. Enclosure 2 provides the changes made to the WBN TRM for this period. Changes to the TS Bases and TRM made only to address pagination or format are not included in this transmittal.

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APR 20 2004

If you have any questions, please contact me at (423) 365-1824.

Sincerely,

P. L. Pace, Manager
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Enclosure
cc (Enclosure):

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ENCLOSURE 1

WBN TECHNICAL SPECIFICATIONS BASES - CHANGED PAGES

# B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

## BASES

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| LCOs | LCO 3.0.1 through LCO 3.0.7 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated. |
| LCO 3.0.1 | LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification). |

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering

(continued)

# B 3.1 REACTIVITY CONTROL SYSTEMS

## B 3.1.5 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," and GDC 26, "Reactivity Control System Redundancy and Capability," (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately 5/8 inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control (Shutdown Banks C and D have only one group each). A group consists

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BACKGROUND (continued)

of two or more RCCAs that are electrically paralleled to step simultaneously. Except for Shutdown Banks C and D, a bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. There are four control banks and four shutdown banks.

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are moved in an overlap pattern, using the following withdrawal sequence: When control bank A reaches a predetermined height in the core, control bank B begins to move out with control bank A. Control bank A stops at the position of maximum withdrawal, and control bank B continues to move out. When control bank B reaches a predetermined height, control bank C begins to move out with control bank B. This sequence continues until control banks A, B, and C are at the fully withdrawn position, and control bank D is approximately halfway withdrawn. The insertion sequence is the opposite of the withdrawal sequence. The control rods are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions.

The axial position of shutdown rods and control rods is indicated by two separate and independent systems, which are the Bank Demand Position Indication System (commonly called group step counters) and the Analog Rod Position Indication (ARPI) System.

The Bank Demand Position Indication System counts the pulses from the rod control system that moves the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± 5/8 inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The ARPI System provides an accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube with a center to center distance of 3.75 inches, which is six

(continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Shutdown Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth, SDM and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability," and GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control (Shutdown Banks C and D have only one group each). A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. Except for Shutdown Banks C and D, a bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. There are four control banks and four shutdown banks. See LCO 3.1.5, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.8, "Rod Position Indication," for position indication requirements.

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally automatically controlled by the Rod Control System, but they can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating). The control banks must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal full power operations.

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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Control Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth, SDM, and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability," and GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control (Shutdown Banks C and D have only one group each). A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. Except for Shutdown Banks C and D, a bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. There are four control banks and four shutdown banks. See LCO 3.1.5, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.8, "Rod Position Indication," for position indication requirements.

The control bank insertion limits are specified in the COLR. An example is provided for information only in Figure B 3.1.7-1. The control banks are required to be at or above the insertion limit lines.

Figure B 3.1.7-1 also indicates how the control banks are moved in an overlap pattern. Overlap is the distance traveled together by two control banks. The predetermined

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## B 3.1 REACTIVITY CONTROL SYSTEMS

## B 3.1.8 Rod Position Indication

### BASES

BACKGROUND

According to GDC 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1.8 is required to ensure OPERABILITY of the control rod position indicators to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The OPERABILITY, including position indication, of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM. Rod position indication is required to assess OPERABILITY and misalignment.

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment and OPERABILITY have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved out of the core (up or withdrawn) or into the core (down or inserted) by their control rod drive mechanisms. The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control (Shutdown Banks C and D have only one group each).

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BASES

APPLICABLE SAFETY ANALYSES (continued)

zero power (HZP) is a process variable that is an initial condition of DBAs, such as the rod cluster control assembly(RCCA) withdrawal, RCCA ejection, and main steam line break accidents performed at zero power that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

All low power safety analyses assume initial RCS loop temperatures ≥ the HZP temperature of 557°F (Ref. 1). The minimum temperature for criticality limitation provides a small band, 6°F, for critical operation below HZP. This band allows critical operation below HZP during plant startup and does not adversely affect any safety analyses since the MTC is not significantly affected by the small temperature difference between HZP and the minimum temperature for criticality.

The RCS minimum temperature for criticality satisfies Criterion 2 of the NRC Policy Statement.

LCO

Compliance with the LCO ensures that the reactor will not be made or maintained critical ($k_{eff} \geq 1.0$) at a temperature less than a small band below the HZP temperature, which is assumed in the safety analysis. Failure to meet the requirements of this LCO may produce initial conditions inconsistent with the initial conditions assumed in the safety analysis.

APPLICABILITY

In MODE 1 and MODE 2, with $k_{eff} \geq 1.0$, LCO 3.4.2 is applicable since the reactor can only be critical ($k_{eff} \geq 1.0$) in these MODES.

The special test exception of LCO 3.1.10, "PHYSICS TESTS Exceptions - MODE 2," permits PHYSICS TESTS to be performed at ≤ 5% RTP with RCS loop average temperatures slightly lower than normally allowed so that fundamental nuclear characteristics of the core can be verified. In order for nuclear characteristics to be accurately measured, it may be necessary to operate outside the normal restrictions of this LCO. For example, to measure the MTC at beginning of cycle, it is necessary to allow RCS loop average temperatures to fall below $T_{no\ load}$, which may cause RCS loop average temperatures to fall below the temperature limit of this LCO.

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# B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

## B 3.5.2 ECCS - Operating

BASES

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system;

b. Rod ejection accident;

c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater; and

d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

There are three phases of ECCS operation: injection, cold leg recirculation, and hot leg recirculation. In the injection phase, water is taken from the refueling water storage tank (RWST) and injected into the Reactor Coolant System (RCS) through the cold legs. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment sumps have enough water to supply the required net positive suction head to the ECCS pumps, suction is switched to the containment sump for cold leg recirculation. Approximately 9 hours after event initiation, the ECCS flow is shifted to the hot leg recirculation phase to provide a backflush, which would reduce the boiling in the top of the core and any resulting boron precipitation.

The ECCS consists of three separate subsystems: centrifugal charging (high head), safety injection (SI) (intermediate head), and residual heat removal (RHR) (low head). Each subsystem consists of two redundant, 100% capacity trains. The ECCS accumulators and the RWST are also part of the

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Misalignment of these valves could render both ECCS trains inoperable. Securing these valves in position by removal of power or by key locking the control in the correct position ensures that they cannot change position as a result of an active failure or be inadvertently misaligned. These valves are of the type, described in Reference 6, that can disable the function of both ECCS trains and invalidate the accident analyses. A 12-hour Frequency is considered reasonable in view of other administrative controls that will ensure a mispositioned valve is unlikely.

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.2.3

With the exception of the operating centrifugal charging pump, the ECCS pumps are normally in a standby, nonoperating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water by venting the ECCS pump casings and accessible suction and discharge piping high points ensures that the system will perform properly, injecting its full capacity into the RCS upon demand.* This will also prevent water hammer, pump cavitation, and pumping of noncondensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SI signal or during shutdown cooling. The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the procedural controls governing system operation. A note is added to the FREQUENCY that surveillance performance is not required for

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SURVEILLANCE REQUIREMENTS

SR 3.5.2.3 (continued)

safety injection hot leg injection lines until startup from the Fall 2003 Refueling Outage. (Ref. 7)

*For the accessible locations, UT may be substituted to demonstrate the piping is full of water. An accessible ECCS high point is defined as one that:

1) Has a vent connection installed.
2) The high point can be vented with the dose received remaining within ALARA expectations. ALARA for venting ECCS high point vents is considered to not be within ALARA expectations when the planned, intended collective dose for the activity is unjustifiably higher than industry norm, or the licensee's past experience, for this (or similar) work activity.
3) The high point can be vented with industrial safety expectations remaining within the industry norm.

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by Section XI of the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pumps baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program, which encompasses Section XI of the ASME Code. Section XI of the ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.5 and 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative control. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF

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# B 3.6 CONTAINMENT SYSTEMS

## B 3.6.4 Containment Pressure

BASES

**BACKGROUND**

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential (-2.0 psid) with respect to the shield building annulus atmosphere in the event of inadvertent actuation of the Containment Spray System or Air Return Fans.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

**APPLICABLE SAFETY ANALYSES**

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer pressure transients. The worst case LOCA generates larger mass and energy release than the worst case SLB. Thus, the LOCA event bounds the SLB event from the containment peak pressure standpoint (Ref. 1).

The initial pressure condition used in the containment analysis was 15.0 psia. This resulted in a maximum peak pressure from a LOCA of 10.64 psig. The containment analysis (Ref. 1) shows that the maximum allowable internal containment pressure, $P_a$ (15.0 psig), bounds the calculated results from the limiting LOCA. The maximum containment pressure resulting from the worst case LOCA, does not exceed the containment design pressure, 13.5 psig.

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BASES

BACKGROUND (continued)

and water from a DBA. During the post blowdown period, the Air Return System (ARS) is automatically started. The ARS returns upper compartment air through the divider barrier to the lower compartment. This serves to equalize pressures in containment and to continue circulating heated air and steam through the ice condenser, where heat is removed by the remaining ice and by the Containment Spray System after the ice has melted.

The Containment Spray System limits the temperature and pressure that could be expected following a DBA. Protection of containment integrity limits leakage of fission product radioactivity from containment to the environment.

APPLICABLE SAFETY ANALYSES

The limiting DBAs considered relative to containment OPERABILITY are the loss of coolant accident (LOCA) and the steam line break (SLB). The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed, in regard to containment ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train of the Containment Spray System, the RHR System, and the ARS being rendered inoperable (Ref. 2).

The DBA analyses show that the maximum peak containment pressure of 10.64 psig results from the LOCA analysis, and is calculated to be less than the containment design pressure. The maximum peak containment atmosphere temperature results from the SLB analysis. The calculated transient containment atmosphere temperatures are acceptable for the DBA SLB.

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BASES

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| APPLICABLE SAFETY ANALYSES (continued) | assumes that radioactive materials leaked from the Emergency Core Cooling System (ECCS) are filtered and adsorbed by the ABGTS. The DBA analysis of the fuel handling accident assumes that only one train of the ABGTS is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the one remaining train of this filtration system. The amount of fission products available for release from the ABSCE is determined for a fuel handling accident and for a LOCA. The assumptions and the analysis for a fuel handling accident follow the guidance provided in Regulatory Guide 1.25 (Ref. 5) and NUREG/CR-5009 (Ref. 11). The assumptions and analysis for a LOCA follow the guidance provided in Regulatory Guide 1.4 (Ref. 6).<br><br>The ABGTS satisfies Criterion 3 of the NRC Policy Statement. |
| LCO | Two independent and redundant trains of the ABGTS are required to be OPERABLE to ensure that at least one train is available, assuming a single failure that disables the other train, coincident with a loss of offsite power. Total system failure could result in the atmospheric release from the ABSCE exceeding the 10 CFR 100 (Ref. 7) limits in the event of a fuel handling accident or LOCA.<br><br>The ABGTS is considered OPERABLE when the individual components necessary to control exposure in the fuel handling building are OPERABLE in both trains. An ABGTS train is considered OPERABLE when its associated:<br><br>a. Fan is OPERABLE;<br><br>b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration function; and<br><br>c. Heater, moisture separator, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained. |
| APPLICABILITY | In MODE 1, 2, 3, or 4, the ABGTS is required to be OPERABLE to provide fission product removal associated with ECCS leaks due to a LOCA and leakage from containment and annulus. |

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REFERENCES (continued)

5. Regulatory Guide 1.25, March 1972, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

6. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors."

7. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

8. Regulatory Guide 1.52 (Rev. 2), "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Power Plants."

9. NUREG-0800, Section 6.5.1, "Standard Review Plan," Rev. 2, "ESF Atmosphere Cleanup System," July 1981.

10. Watts Bar Drawing 1-47W605-242, "Electrical Tech Spec Compliance Tables."

11. NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," U. S. Nuclear Regulatory Commission, February 1988.

BASES

ACTIONS

B.2 and C.2 (continued)

b. A required feature on the other train (Train A or Train B) is inoperable.

If at any time during the existence of this Condition (one or more DGs inoperable) a required feature subsequently becomes inoperable, this Completion Time would begin to be tracked.

Discovering one or more required DGs in Train A or one or more DGs in Train B inoperable coincident with one or more inoperable required support or supported features, or both, that are associated with the OPERABLE DGs, results in starting the Completion Time for the Required Action. Four hours from the discovery of these events existing concurrently is Acceptable because it minimizes risk while allowing time for restoration before subjecting the plant to transients associated with shutdown.

In this Condition, the remaining OPERABLE DGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection for the required feature's function may have been lost; however, function has not been lost. The 4 hour Completion Time takes into account the OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

B.3.1, B.3.2, C.3.1 and C.3.2

Required Actions B.3.1 and C.3.1 provide an allowance to avoid unnecessary testing of OPERABLE DG(s). If it can be determined that the cause of the inoperable DG does not exist on the OPERABLE DG, SR 3.8.1.2 does not have to be performed. For the performance of a Surveillance, Required Action B.3.1 is considered satisfied since the cause of the DG being inoperable is apparent. If the cause of inoperability exists on other DG(s), the other DG(s) would be declared inoperable upon discovery and Condition F of LCO 3.8.1 would be entered if the other inoperable DGs are not on the same train, otherwise, if the other inoperable DGs are on the same train, the unit is in Condition C. Once

(continued)

BASES

ACTIONS

B.3.1, B.3.2, C.3.1 and C.3.2 (continued)

the failure is repaired, the common cause failure no longer exists, and Required Actions B.3.1 and B.3.2 are satisfied. If the cause of the initial inoperable DG cannot be confirmed not to exist on the remaining DG(s), performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of that DG.

In the event the inoperable DG is restored to OPERABLE status prior to completing either B.3.1, B.3.2, C.3.1 or C.3.2, the corrective action program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 12 hour constraint imposed while in Condition B or C.

B.4

In Condition B, the remaining OPERABLE DGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. The 14 day Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action B.4 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an offsite circuit is inoperable and that circuit is subsequently restored OPERABLE, the LCO may already have been not met for up to 14 days. This could lead to a total of 17 days, since initial failure to meet the LCO, to restore the DGs. At this time, an offsite circuit could again become inoperable, the DGs restored OPERABLE, and an additional 72 hours (for a total of 20 days) allowed prior to complete restoration of the LCO. The 17 day Completion Time provides a limit on time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The "AND" connector between the 14 day and 17 day Completion Times means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

(continued)

ACTIONS

B.4 (continued)

Compliance with the contingency actions listed in Bases Table 3.8.1-2 is required whenever Condition B is entered for a planned or unplanned outage which will extend beyond 72 hours. If Condition B is entered initially for an activity intended to last less than 72 hours or for an unplanned outage, the contingency actions should be invoked as soon as it is established that the outage period will be longer than 72 hours. The contingency actions applicable to Surveillance Requirement (SR) 3.8.1.14 must be invoked prior to initiation of the test.

As in Required Action B.2, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed time "clock." This will result in establishing the "time zero" at the time that the LCO was initially not met, instead of at the time Condition B was entered.

According to TVA's probabilistic safety analysis described in Reference 11, 12 hours is reasonable to confirm the OPERABLE DGs are not affected by the same problem as the inoperable DG.

C.4

According to Regulatory Guide 1.93, (Ref. 6), operation may continue in Condition C for a period that should not exceed 72 hours.

In Condition C, the remaining OPERABLE DGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period. Restoration of at least one DG within 72 hours results in reverting back under Condition B and continuing to track the "time zero" completion time for one DG inoperable.

The second Completion Time for Required Action C.4 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition C is entered while, for instance, an offsite circuit is inoperable and that circuit is subsequently restored OPERABLE, the LCO may already have been not met for up to 72 hours. This could lead to a total of 144 hours, since initial failure to meet the LCO, to restore the DGs. At this time, an offsite circuit could again become inoperable, the DGs restored OPERABLE, and an additional 72 hours (for a total of 9 days) allowed prior to

(continued)

**Bases Table 3.8.1-2**
**TS Action or Surveillance Requirement (SR) Contingency Actions**

| | Contingency Actions to be Implemented | Applicable TS Action or SR | Applicable Modes |
|---|---|---|---|
| 1. | Verify that the offsite power system is stable. This action will establish that the offsite power system is within single-contingency limits and will remain stable upon the loss of any single component supporting the system. If a grid stability problem exists, the planned DG outage will not be scheduled. | SR 3.8.1.14<br>Action B.4 | 1, 2<br>1, 2, 3, 4 |
| 2. | Verify that no adverse weather conditions are expected during the outage period. The planned DG outage will be postponed if inclement weather (such as severe thunderstorms or heavy snowfall) is projected. | SR 3.8.1.14<br>Action B.4 | 1, 2<br>1, 2, 3, 4 |
| 3. | Do not remove from service the ventilation systems for the 6.9 kV shutdown board room, the elevation 772 transformer room, or the Unit 2 480-volt shutdown board room, concurrently with the DG, or implement appropriate compensatory measures. | Action B.4 | 1, 2, 3, 4 |
| 4. | Do not remove the reactor trip breakers from service concurrently during planned DG outage maintenance. | Action B.4 | 1, 2, 3, 4 |
| 5. | Do not remove the turbine-driven auxiliary feedwater (AFW) pump from service concurrently with a Unit 1 DG outage. | Action B.4 | 1, 2, 3, 4 |
| 6. | Do not remove the AFW level control valves to the steam generators from service concurrently with a Unit 1 DG outage. | Action B.4 | 1, 2, 3, 4 |
| 7. | Do not remove the opposite train residual heat removal (RHR) pump from service concurrently with a Unit 1 DG outage. | Action B.4 | 1, 2, 3, 4 |

BASES

ACTIONS (continued)

B.1

With lube oil inventory < 287 gal per diesel engine, sufficient lubricating oil to support 7 days of continuous DG operation at full load conditions may not be available. However, the Condition is restricted to lube oil volume reductions that maintain at least a 6 day supply. This restriction allows sufficient time to obtain the requisite replacement volume. A period of 48 hours is considered sufficient to complete restoration of the required volume prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (> 6 days), the low rate of usage, the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

C.1

This Condition is entered as a result of a failure to meet the acceptance criterion of SR 3.8.3.3. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, and particulate concentration is unlikely to change significantly between Surveillance Frequency intervals, and proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG inoperable. The 7 day Completion Time allows for further evaluation, resampling and re-analysis of the DG fuel oil.

D.1

With the new fuel oil properties defined in the Bases for SR 3.8.3.3 not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or to restore the stored fuel oil properties. This

(continued)

ACTIONS

D.1 (continued)

restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the DG would still be capable of performing its intended function.

E.1

With starting air receiver pressure < 190 psig, sufficient capacity for five successive DG start attempts does not exist. However, as long as the receiver pressure is ≥ 170 psig (value does not account for instrument error, Ref. 7), there is adequate capacity for at least one start attempt, and the DG can be considered OPERABLE while the air receiver pressure is restored to the required limit of ≥ 190 psig (value does not account for instrument error, Ref. 7). A period of 48 hours is considered sufficient to complete restoration to the required pressure prior to declaring the DG inoperable. This period is acceptable based on the remaining air start capacity, the fact that most DG starts are accomplished on the first attempt, and the low probability of an event during this brief period.

F.1

With a Required Action and associated Completion Time not met, or one or more DG's fuel oil, lube oil or starting air subsystem not within limits for reasons other than addressed by Conditions A through E, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory (≥ 56,754 gallons, value does not account for instrument error, Ref. 7) of fuel oil in the storage tanks to support each DG's operation for 7 days at full load. The 7 day period is sufficient time to place the plant in a safe shutdown condition and to bring in replenishment fuel from an offsite location.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.4.5 and SR 3.8.4.6 (continued)

The limits established for this SR must be no more than 20% above the resistance as measured during installation, or not above the ceiling value established by the manufacturer.

The Surveillance Frequency for these inspections, which can detect conditions that can cause power losses due to resistance heating, is 92 days. This Frequency is considered acceptable based on operating experience related to detecting corrosion trends.

SR 3.8.4.7

Visual inspection of the battery cells, cell plates, and battery racks provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance.

The 12 month Frequency for this SR is consistent with IEEE-450 (Ref. 9), which recommends detailed visual inspection of cell condition and rack integrity on a yearly basis.

SR 3.8.4.8, SR 3.8.4.9 and SR 3.8.4.10

Visual inspection and resistance measurements of intercell, interrack, intertier, and terminal connections provide an indication of physical damage or abnormal deterioration that could indicate degraded battery condition. The anticorrosion material is used to help ensure good electrical connections and to reduce terminal deterioration. The visual inspection for corrosion is not intended to require removal of and inspection under each terminal connection. The removal of visible corrosion is a preventive maintenance SR. The presence of visible corrosion does not necessarily represent a failure of this SR provided visible corrosion is removed during performance of SR 3.8.4.8. For the purposes of trending, inter-cell (vital and DG batteries) and inter-tier (vital batteries) connections are measured from battery post to battery post. Inter-rack (vital and DG batteries), inter-tier (DG Batteries), and terminal connections (vital and DG batteries) are measured from terminal lug to battery post.

(continued)

# B 3.9 REFUELING OPERATIONS

## B 3.9.7 Refueling Cavity Water Level

BASES

**BACKGROUND**

The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to < 25% of 10 CFR 100 limits, as provided by the guidance of Reference 3.

**APPLICABLE SAFETY ANALYSES**

During movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1) except for I-131 which is assumed to be 12% (Ref. 6).

The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 100 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Refs. 4 and 5).

Refueling cavity water level satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel-Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, March 23, 1972.

2. Watts Bar FSAR, Section 15.4.5, "Fuel Handling Accident."

3. NUREG-0800, "Standard Review Plan," Section 15.7.4, "Radiological Consequences of Fuel-Handling Accidents," U.S. Nuclear Regulatory Commission.

4. Title 10, Code of Federal Regulations, Part 20.1201(a), (a)(1), and(2)(2), "Occupational Dose Limits for Adults."

5. Malinowski, D. D., Bell, M. J., Duhn, E., and Locante, J., WCAP-7828, Radiological Consequences of a Fuel Handling Accident, December 1971.

6. NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," U. S. Nuclear Regulatory Commission, February 1988.

ENCLOSURE 2

WBN TECHNICAL REQUIREMENTS MANUAL - CHANGED PAGES

1.1 Definitions (continued)

| | |
|---|---|
| QUADRANT POWER TILT RATIO (QPTR) | QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average lower excore detector calibrated outputs, whichever is greater. |
| RATED THERMAL POWER (RTP) | RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3459 MWt. |
| REACTOR TRIP SYSTEM (RTS) RESPONSE TIME | The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC. |
| SHUTDOWN MARGIN (SDM) | SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:<br>a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and<br>b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level. |

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

-----------------------------------NOTES-----------------------------------

1. TSR 3.1.5.1, TSR 3.1.5.2 and TSR 3.1.5.3 are only required to be performed if the RWST is the required borated water source.

2. TSR 3.1.5.4, TSR 3.1.5.5 and TSR 3.1.5.6 are only required to be performed if the Boric Acid Storage System is the required borated water source.

---------------------------------------------------------------------------

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| TSR 3.1.5.1 | ------------------NOTE------------------<br>Only required when ambient air temperature is < 60°F.<br>----------------------------------------<br><br>Verify RWST solution temperature is ≥ 60°F. | <br><br><br><br>24 hours |
| TSR 3.1.5.2 | Verify RWST boron concentration is ≥ 3,100 ppm. | 7 days |
| TSR 3.1.5.3 | Verify RWST borated water volume is ≥ 62,900 gallons. | 7 days |
| TSR 3.1.5.4 | Verify Boric Acid Tank (BAT) solution temperature is ≥ 63°F. | 24 hours |
| TSR 3.1.5.5 | Verify BAT boron concentration is ≥ 6,120 and ≤ 6,990 ppm. | 7 days |
| TSR 3.1.5.6 | Verify BAT borated water volume is ≥ 5,100 gallons. | 7 days |

ACTIONS (continued)

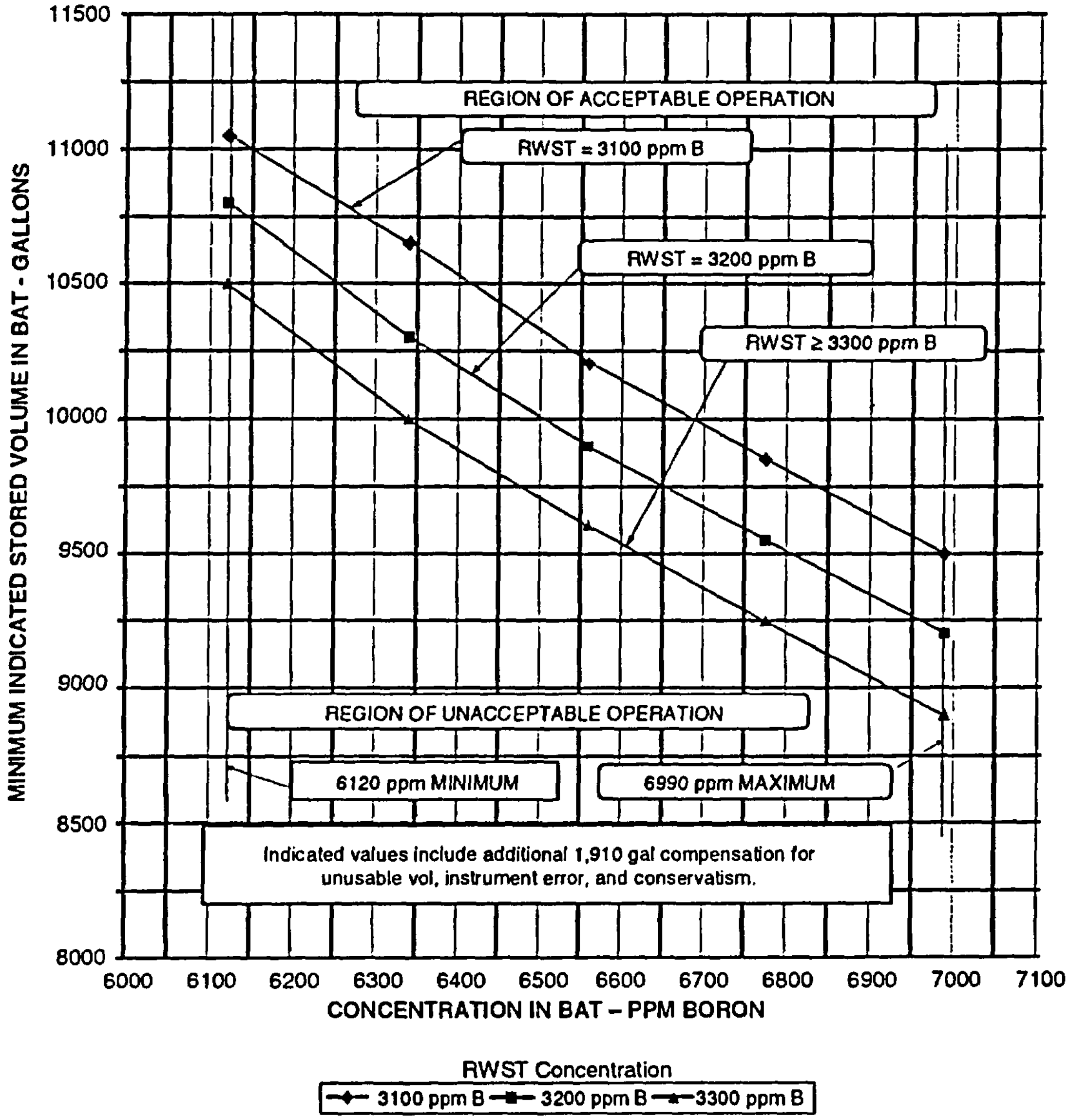
| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| C. RWST boron concentration not within limits.<br><br>OR<br><br>RWST borated water temperature not within limits. | C.1 Restore RWST to OPERABLE status. | 8 hours |
| D. RWST inoperable for reasons other than Condition C. | D.1 Restore RWST to OPERABLE status. | 1 hour |
| E. Required Action and associated Completion Time of Condition C or D not met. | E.1 Be in MODE 3<br><br>AND<br><br>E.2 Be in MODE 4 with one or more RCS cold leg temperatures ≤ 310 °F. | 6 hours<br><br><br><br>12 hours |

TECHNICAL SURVEILLANCE REQUIREMENTS

| | SURVEILLANCE | FREQUENCY |
|---|---|---|
| TSR 3.1.6.1 | -----------------NOTE------------------<br>Only required when outside air temperature is < 60 °F or >105 °F.<br>---------------------------------------<br><br>Verify RWST solution temperature is ≥ 60° F and ≤ 105 °F. | <br><br><br><br>24 hours |
| TSR 3.1.6.2 | Verify RWST boron concentration is ≥ 3,100 ppm and ≤ 3,300 ppm. | 7 days |

(continued)

TECHNICAL REQUIREMENTS FIGURE 3.1.6
BORIC ACID TANK LIMITS
BASED ON RWST BORON CONCENTRATION
For 0 to 240 TPBARS



BASES

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.1.5.1 (continued)

is greater than or equal to 60°F. With ambient air temperature greater than 60°F, the RWST solution temperature should not decrease below this limit, therefore, monitoring is not required.

TSR 3.1.5.2

This surveillance requires verification every 7 days that the boron concentration of the RWST is ≥ 2,500 ppm. This boron concentration is sufficient to provide an adequate SDM and also ensure a pH value between 7.5 and 10.0. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. Since the RWST volume is normally stable, a 7-day Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

TSR 3.1.5.3

This surveillance requires verification every 7 days that the RWST borated water volume is ≥ 62,900 gallons (value does not account for instrument error). This borated water volume is sufficient to provide an adequate SDM and also ensure a pH value between 7.5 and 10.0. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. Since the RWST volume is normally stable, a 7-day Frequency to verify borated water volume is appropriate and has been shown to be acceptable through operating experience. The 62,900 gallon volume requirement includes 11,100 gallons for shutdown margin and adjustments for minimum safety limit level in the RWST.

TSR 3.1.5.4

This surveillance requires verification every 24 hours that the Boric Acid Tank (BAT) solution temperature is ≥ 63°F (value does not account for instrument error). This ensures that the concentration of boric acid in the BAT is not allowed to precipitate due to cooling. The Frequency of 24 hours for performance of the surveillance is frequent enough to identify a temperature change that would approach the 63°F temperature limit.

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.1.5.1 (continued)

is greater than or equal to 60°F. With ambient air temperature greater than 60°F, the RWST solution temperature should not decrease below this limit, therefore, monitoring is not required.

TSR 3.1.5.2

This surveillance requires verification every 7 days that the boron concentration of the RWST is ≥ 3,100 ppm. This boron concentration is sufficient to provide an adequate SDM and also ensure a pH value between 7.5 and 10.0. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. Since the RWST volume is normally stable, a 7-day Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

TSR 3.1.5.3

This surveillance requires verification every 7 days that the RWST borated water volume is ≥ 62,900 gallons (value does not account for instrument error). This borated water volume is sufficient to provide an adequate SDM and also ensure a pH value between 7.5 and 10.0. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. Since the RWST volume is normally stable, a 7-day Frequency to verify borated water volume is appropriate and has been shown to be acceptable through operating experience. The 62,900 gallon volume requirement includes 10,900 gallons for shutdown margin and adjustments for minimum safety limit level in the RWST.

TSR 3.1.5.4

This surveillance requires verification every 24 hours that the Boric Acid Tank (BAT) solution temperature is ≥ 63°F (value does not account for instrument error). This ensures that the concentration of boric acid in the BAT is not allowed to precipitate due to cooling. The Frequency of 24 hours for performance of the surveillance is frequent enough to identify a temperature change that would approach the 63°F temperature limit.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.1.5.5

This surveillance requires verification every 7 days that the boron concentration of the BAT is between 6,120 ppm and 6,990 ppm. This boron concentration is sufficient to provide an adequate SDM and also ensure a pH value between 7.5 and 10.0. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. Since the BAT volume is normally stable, a 7-day Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

TSR 3.1.5.6

This surveillance requires verification every 7 days that the BAT borated water volume is ≥ 5,100 gallons (value does account for instrument error). This borated water volume is sufficient to provide an adequate SDM and also ensure a pH value between 7.5 and 10.0. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. Since the BAT volume is normally stable, a 7-day Frequency to verify borated water volume is appropriate and has been shown to be acceptable through operating experience.

REFERENCES

1. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

2. CEN-603, "Boric Acid Concentration Reduction Effort, Technical Bases and Operational Analysis for Watts Bar, Unit 1," Revision 00, April 1993.

3. TVA Calculation, EPM-PDM-071197, Revision 3, "Boric Acid Concentration Analysis for BAT and RWST."

4. Westinghouse Letter, WAT-D-10940, Revision 1, "Watts Bar Unit One TRM & FSAR Markups - Post LOCA Sump pH."

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.1.6.5 (continued)

7-day sampling Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

This surveillance has been modified by a NOTE stating that the surveillance is only required if the BAT is used as one of the required borated water sources for TR 3.1.2.

TSR 3.1.6.6

This surveillance requires verification every 7 days that the BAT borated water volume is in accordance with Figure 3.1.6 (the values listed on the figure do not account for instrument error). This borated water volume at the boron concentration specified in TSR 3.1.6.5 is sufficient to provide an adequate SDM. Since the BAT volume is normally stable, a 7-day Frequency to verify borated water volume is appropriate and has been shown to be acceptable through operating experience.

This surveillance has been modified by a NOTE stating that the surveillance is only required if the BAT is used as one of the required borated water sources for TR 3.1.2.

The maximum expected boration capability requirement occurs near EOL from full power peak xenon conditions and requires borated water from a boric acid tank in accordance with Figure 3.1.6, and additional makeup from either (1) the common boric acid tank and/or batching tank, or (2) a maximum of 22,000 gallons of 3,100 ppm borated water from the refueling water storage tank. With the refueling water storage tank as the only borated water source, a maximum of 61,000 gallons of 3,100 ppm borated water is required.

REFERENCES

1. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

2. CEN-603, "Boric Acid Concentration Reduction Effort, Technical Bases and Operational Analysis for Watts Bar Nuclear Plant, Unit 1," Revision 00, April 1993.

3. TVA Calculation, EPM-PDM-071197, Revision 3, "Boric Acid Concentration Analysis For BAT and RWST."

4. Westinghouse Letter, WAT-D-10940, Revision 1, "Watts Bar Unit One TRM and FSAR Markups - Post LOCA Sump pH."